Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/P/306
USSR
May 1964
Original: RUSSIAN

Confidential until official release during Conference

EXPERIMENTAL PRESSURE VESSEL REACTOR FOR STUDIES IN BOILING AND STEAM SUPERHEATING

I.N.Sokolov, V.I.Gritskov, P.N.Bogdanovich, E.V.Kulikov, S.A.Skvortsov, V.P.Spasskov, V.I.Plutinsky, I.A.Philatyev, A.A.Hohlachev.

EXPERIMENTAL PLANT PURPOSES

Nuclear boiling water reactors and boiling reactors with nuclear superheat attract a great attention of nuclear engineers as a possible promissing trend in nuclear power engineering development.

Though there is no complete assurance that the development in this way will substantially improve technical and economic characteristics of water-cooled and moderated reactors, in the Soviet Union an extensive programme for studying boiling reactors including reactors with nuclear superheating of steam is to be carried out.

This year in the Ulyanovsk region an experimental power plant for the studies in boiling and steam superheating problems is being completed.

The reactor of the plant under construction is designed for 30 to 100 atm steam generation. This will allow to investigate the boiling effects on physical characteristics in a wide pressure range. As practically the working steam pressure in a power plant is chosen according to its output so the experimental plant under construction will enable to obtain data necessary for designing large and small plants.

The reactor design envisages investigations of various cores. Firstly the performance of two cores is to be studied. At equal maximum specific loads a "small" core (1.8 m diameter x 2 m high) is designed for production of 100 atm saturated steam at a power level of 150 Mw(t). A "large" core (2.6 m diameter x 2 m high) is designed for generating 90 atm superheated steam at a temperature from 500 to 510°C at a higher power level. In future superheated steam temperature will be raised to 565 – 580°C.

The plant equipment is designed for extraction of 630 t/hr steam which may be sent either to the turbines (total electric power about 90 Mw) or to an auxiliary condenser.

In addition to the studies of physical and operational reactor characteristics the plant will enable the reliability of the approved electric power circuits and their separate components to be proved under operating conditions. In particular, the reliability and safety of the automatic control system of the plant as a whole must be examined. The circulating water

purification system efficiency, fuel element life during long operation in steam-water mixture and superheated steam media, H_2 and O_2 release from the reactor under various conditions and other problems will be solved during the test operation of the plant under construction.

HEAT REMOVAL SYSTEM

A. SATURATED STEAM GENERATION CIRCUIT

As it has been pointed out steam at different pressures can be generated in the reactor. The turbine is of the type as those of the Novo-Voronezh power station with the BBЭP reactor (water-water power reactor). It is designed for 29 atm saturated steam (2). Because of this, the heat removal system envisages steam throttling before it goes to the turbine when the reactor operates at a pressure higher than 50 atm.

The plant flow diagram is shown in Fig. 1.

Saturated steam is produced in the reactor (1). Primary steam separation takes place in the reactor vessel. The steam with moisture content not more than 1% enters the high-pressure separator (2) and then passes the throttle valve (3) where the pressure reduces to 30 atm, the moisture content being increased. A mean pressure separator is provided to dry the steam.

The turbine, which the steam enters through the mean pressure separator, has two cylinders, a moisture separator being installed between them (6).

The condensate is fed by feed water pumps from the deaerator either to the reactor or to the reactor and steam generators (9). Some water passes through the shrouds of the excess reactivity compensation system mounted on the top of the reactor vessel to prevent the drives from overheating due to steam leakage from the reactor into them.

At the plant it is envisaged to install three steam generators, where 30 atm saturated steam is produced by cooling water removed from the reactor. Each steam generator is included in the loop provided with a pump with a capacity of 500 t/hr. The steam generated in the steam generators is delivered to the turbine steam main.

The steam generators are included in the circuit to test the operational characteristics of the duel cycle, where the plant output may be increased due to steam production in the steam generators. In addition, the steam generators allow the turbine adjustment during the plant start-up using nonradioactive steam, in this case the steam generated in the reactor is dumped into the condenser (11).

Simultaneous steam production in the reactor and the steam generators may be carried out using successive water feeding according to the scheme of so-called step evaporation proposed by professor Romm E.I. (3). In this case the steam generators are fed with water removed from the reactor. It makes low salt concentration of primary water possible as a result of substantial increase in the amount of water removed from the reactor. The plant circuit envisages step evaporation.

As the plant under construction is considered experimental the reactor may operate without the turbine [redacted]. For such a case an auxiliary condenser (10) is provided in the circuit to by-pass steam into.

B. SUPERHEATED STEAM GENERATION CIRCUIT

100 atm saturated steam is produced in the "large" core boiling zone which is several times larger than the "small" core. Steam moisture separation being done, the steam goes from the vapour plenum of the reactor vessel to superheating fuel assemblies which are at the periphery of the "small" core. In each assembly of the superheater zone the steam is superheated in two passes in such a way as the steam at a pressure of 90 atm and superheat temperature of 510°C is again delivered to the upper part of the reactor vessel and through special manifolds to the steam nozzles of the reactor vessel. The superheated steam is delivered through the steam lines either to a 25 Mw single cylinder Vorschaltturbine or a special cooler. 30 atm steam is delivered from Vorschaltturbine or the cooler to the turbine similar to that of the Novo-Voronezh nuclear power station.

This scheme makes it possible to generate saturated steam either in the reactor vessel or in the steam generators and the reactor. If steam is generated in the steam generators and the reactor then superheated steam having gone through Vorschaltturbine is mixed with saturated steam in a special mixer, the latter being generated in the steam generators.

The designed reactor power is 250 Mw(t), its steam output at a pressure of 100 atm is 400 t/hr and the total steam generators output is 210 t/hr (4). The turbines use 400 t/hr steam at full load and the auxiliary condenser is designed for passing 250 t/hr steam. The plant can operate with various schemes of switching steam producing and steam consuming equipment.

REACTOR

The reactor core, 2.6 m diameter and 2 m high, is located within a steel vessel in the form of a vertical cylinder vessel, 3.8 m outer diameter and 11 m high (Figs 2 and 3).

Assemblies of 10.2 mm rod-type fuel elements, are within the core. The fuel is in the form of pellets of sintered uranium dioxide enriched to 1.5 – 2.0% in the boiling zone and to 3.0 – 3.5% in the superheater one. In the boiling zone the pellets are encased in a 0.6 mm wall tubular cladding of zirconium alloy, the cladding of the superheating fuel elements is of special heat-resistant 0.5 mm stainless steel.

127 fuel elements in a 15.1 mm spacing triangular lattice are encased in a hexahedral in cross section tube of zirconium alloy, forming an assembly of the superheater zone (Fig. 4a).

The cross section size is 176 mm. There are support grids at the top and the bottom of the assemblies. Along the length of the fuel elements the spacing is maintained by special spacer grids to prevent the fuel elements from contact with each other. The ends of the fuel assemblies are provided with end closures the assemblies are entrapped by while refueling. The fuel elements are arranged symmetrically relative to the mean cross section thus they may be inserted any end forward into the core. This makes it possible to turn them over while refueling to achieve a higher burn-up.

At the bottom the assemblies are rested on a steel plate with holes the lower end closures of the assemblies are inserted through. At the top the spacing of the assemblies is maintained by a spacer grid. There is a 9 mm water gap among the assemblies.

306

The fuel assemblies of the superheater zone being of the same size as those of the boiling zone consist of a fewer number of fuel elements, each of them in a heat-resistant stainless steel can is surrounded by a 1 mm gap for superheated steam flowing. There are two steam passes in each fuel assembly, from the reactor steam plenum the saturated steam flows to the peripheral fuel elements of the assembly, at the bottom of the fuel assembly partially superheated steam enters the central region of the assembly where it is superheated to the given temperature.

To reduce heat leakage from the superheated steam to the moderator, a circular steam gap is provided around each fuel element. The spacing of the fuel element in the channel is supposed to be maintained by means of helical fins.

The general view of a superheating fuel element channel is shown in Fig. 4b.

The fuel arrangement in the form of bundles of concentric, plate or rod type fuel elements is envisaged for other superheater zone geometries.

The fuel assemblies arrangement in "large" and "small" cores is shown in Fig. 5. Both of them are arranged concentrically.

The whole core is mounted in a cylindrical basket, the fuel assemblies support plate being the bottom of the basket.

The control and safety assemblies are also shown in this figure. The physical characteristics of both zones are listed in Table 1.

To compensate reactivity 31 assemblies are used, their upper part being provided with a boron steel absorber hexahedral in cross section. These assemblies can move along the reactor vertical axis. While moving the assemblies downward the absorber is inserted into the core and reactivity decreases. While moving the assemblies upward the replacement of the absorber by fuel elements caus[illegible] increase in reactivity. In the core assemblies are arranged in 370 mm spacing triangular l[illegible]e.

6 assemblies containing a zirconium scatterer in the lower section and an absorber in the upper one are used for safety system purposes. During the reactor operation the scatterer is in the core and the absorber is somewhat higher. In case of emergency the assemblies drop downward and as a result in the core the scatterer is replaced by the absorber. These assemblies are made circular in cross section to avoid jamming while their quick downward movement.

The reactor vessel is made of heat-resistant low alloy steel with yield strength of 45 kg/mm^2 at 325°C. An elliptic bottom is welded to the cylindrical part of the vessel.

The vessel inner surface in contact with the coolant is clad with stainless steel. At the top the vessel is covered with a removable head.

The vessel upper ring is used as a flange provided with 60 studs to fix the pressure ring which holds the head in place. The studs are tightened up by a hydraulic spanner. The 500 mm thick head rests on the vessel flange lug, a wedge nickel gasket seals the head. The stand-by sealing is envisaged in the form of a toroidal compensator welded to the head. There are holes in the head for the rods of the control mechanisms drives to pass through.

REACTOR CORES CHARACTERISTICS

Table I

Name	Units	"Large" core	"Small" core
Core dimensions:			
Diameter	m	2.6	1.8
Height	m	2.0	2.0
No. of fuel assemblies in the core (Boiling zone/superheater zone)	–	109/72	85
UO_2 loading in the core (Boiling zone/superheater zone)	t	14.6/4.7	11.4
Water-to-uranium ratio (Boiling zone/superheater zone)	–	2.3/4.0	2.3
Uranium enrichment (Boiling zone/superheater zone)	%	$\frac{1.5-2.0}{3.0-3.5}$	2.0
Balance of reactivity			
a) Temperature effect (to 309°C)	% R eff	2.4	3.5
b) Steam generation (for average steam fraction)	– " –	2.0	up to 3.5
c) Doppler effect	– " –	1.6	2.0
d) Xenon and samarium poisoning (in equilibrium state)	– " –	4.2	4.7
e) Fuel burn-up	– " –	5.7	5.3
f) Structural elements	– " –	1.0	2.5
Total reactivity excess in a cold clean reactor	– " –	17.0	21.5
Worth of control devices in a cold reactor	% K eff	39.0	44.0
including:			
Absorbers with mechanical drives	% K eff	19.0	24.0
Liquid poison	– " –	20.0	20.0
Max. fuel element heat flux	$kcal/m^2hr$	$1.2x10^6$	$1.2x10^6$

Control and safety system mechanisms are located in vertical leak tight shrouds welded to the reactor vessel head. In the upper shroud sockets low frequency three-phase synchronous pole motors are located. Rotation speed is controlled by means of changes in voltage frequency fed to the stator windings. The electric motor is reversed by changing the order of sequences of voltage phases.

A hollow screw is hinged to the lower part of the electric motor rotor, a nut translating up and down the screw when in rotation. In its turn the nut is connected with the rod. There is a collet grip at the end of the rod, the grip releases when positioned in the fuel assembly socket. Thus assemblies and control rods are attached to the drive mechanism.

In case of jamming some control mechanism in the upper position under scram conditions liquid poison is to be injected into the reactor vessel. Liquid poison injection will be also necessary for studies of cores with fuel enrichment to 4 — 5%. The calculated poison concentration has been chosen thus that negative reactivity is equal to 20%.

The reactor is a natural coolant circulation reactor. While operating under boiling conditions the steam-water mixture generated in the fuel assemblies enters the upper top of the

306

reactor chimney and then leaves through the penetrations near the top of the chimney and four steam pipes, 300 mm in diameter.

While operating at steam superheating regime saturated steam enters the fuel assemblies of the superheater zone first and then it goes through three outlet steam pipes. Through one steam pipe saturated steam may leave the reactor by-passing the superheater.

The upper part of the 3.5 m high reactor chimney, filled up with steam-water mixture provides the main circulation.

Through the lower penetrations the water leaves the reactor chimney and flows down to the basket bottom through the annular channel between the reactor chimney and the vessel walls. While passing through the penetrations the water may entrain 10 – 20% steam of the total amount generated in the reactor. The entrained steam may substantially impair the water circulation conditions. To condence the steam feed water is piped into the reactor vessel below the steam-water interface.

The designed circulation velocity is 1.0 m/sec at full power and a pressure of 100 atm. The maximum steam velocity in the fuel assemblies of the superheater zone may be 40-50 m/sec and 65 – 70 m/sec at the power level above nominal.

The water is removed to the steam generators through three nozzles, 200 mm in dia. There are three 300-mm diameter nozzles, in the same vessel portion for the water to flow back from the steam generators. The lower nozzles level ensures the core to be filled up with water in case of steam pipe leakage.

The fuel assemblies are refueled and rearranged, after the reactor shut-down cooling and removing the head by means of a special bridge, the refueling rods being placed over the fuel assemblies by means of a coordinate electric circuit.

Fuel element bundles were successfully tested in loops under designed operational conditions. Steam distribution in the upstream flow section, steam entrainment into the downstream flow section, feed water distribution system were studied at special stands.

AUTOMATIC CONTROL SYSTEM OF THE PLANT

The control system is to adjust the operation of the heat generating and heat consuming parts of the plant to maintain the main plant parameters with given accuracy. The plant control system with the turbine operating at saturated steam is shown in Fig. 6.

Control rod and assembly motion pulse is sent when there is difference between the reactor output characterised by ion chamber current (1) and the power regulator setting (2). A command regulator (3) of steam pressure before the turbine is provided to maintain the amount of steam generated in the reactor equal to amount used in the plant. The steam pressure in a steam pipe changed, the pressure regulator, through a switch (4) set by the operator in some position, may operate either the reactor power regulator (2), or the regulator of the steam dumping into the auxiliary condenser or the turbine rate regulator. In the last case mentioned the turbine output is completely dependent on the reactor output. In addition, the regulator (3) may operate the regulator of the circulating water flow rate through the steam generators (7). In this case the reactor power regulator is switched off and reactifity changes due to the reactor water

306

temperature and steam content changes. The reactor output change results in the change of the steam generated in the reactor ratio to that in the steam generators.

During operation the reactor pressure is maintained constant by means of the regulator (12) changing the position of four valves (16) installed in steam lines (in the scheme they are conventionally shown by one line). If the reactor operates at a pressure of 30 atm, these valves are open and the reactor pressure is maintained by the regulator (3).

The reactor and the steam generators are fed with water by means of the regulators (9 and 10), which are actuated by pulses sent by level indicators and steam and water flow rate meters. As it is necessary to ensure a wide range of the plant load and pressure changes the feed pumps (13) feeding water to the reactor are provided with ~~fluid flywheels~~ hydraulic couplings. The pump number of revolutions is controlled by the ~~fluid flywheel~~ hydraulic coupling through the regulator (11) feeding oil into the ~~fluid flywheel~~ coupling in such a way as to keep the pressure drop across the feeding valve (17) constant.

During the reactor operation with nuclear superheating and an additional ~~Vorschaltturbine~~ topping turbine to adjust the plant steam output to load a master command superheated steam pressure regulator is provided, which depending on the regime given may operate either the reactor power or the turbine rate regulator demands.

The designed automatic control system of the plant was studied by means of an electronic analogue computer with actual regulators. These studies showed that the reactor operation is the most stable when the regulator (3) controls the steam flow rate fed to the auxiliary condenser or the turbine. The former case, however, is not economic and the latter may be accepted provided the nuclear power plant is included into a powerful electric network.

When the regulator (3) operates the heat generating part of the plant (this is most important for independent power stations) simultaneous steam generation in the reactor and the steam generators is the most advantageous. The most smooth control is when the regulator (3) acts on the circulating water flow rate through the steam generators. When the regulator (3) operates the reactor power regulator, the plant transients are a bit larger though in permissible limits.

In case there is no throttle valve between the reactor and the turbine (that is the reactor pressure depends on the pressure before the turbine) (3) a good quality of control is assured when the regulator (3) operates the power regulator (2).

EXPERIMENTAL STUDIES

The experimental plant is designed for studies of neutron physics, heat transfer and fluid flow of boiling reactors and those with nuclear superheat as well as heat transfer and auxiliary equipment and the plant characteristics under steady-state, transient and scram conditions.

The test programme on the determination of main processes and the parameters charactering them is discussed below.

1. PLANT STATICS STUDIES

In the reactor vessel ~~various cores will be~~ with various arrangements of the active zone arranged to study their sub-criticality, to determine differential and integral excess reactivity compensation mechanism worth by the water

306

level change in the reactor vessel, by homogeneous moderator "poisoning" with liquid poison and by pulse source methods.

Studies and measurements of physical effects of reactivity change due to the coolant heating-up, steam generation, Doppler effect, core structural components will be carried out for every new core. Control rod calibration and reactivity pile oscillator techniques are supposed to be used. The pile oscillator technique will be used to determine neutron life time and delayed neutron fraction.

Ion and fission chambers as well as an activation techniques are to be used to measure power density distribution in the reactor core volume.

Fuel burn-up is supposed to be studied by means of spent fuel element assemblies "weighing" at a special stand and calculating integral power density in different core volumes. To study core thermal and hydraulic characteristics quick response pressure, temperature, rate and steam void sensors are positioned in the reactor vessel and directly in the core. They may be used for static as well as dynamic experiments.

The main characteristics with indication of the sensor position to measure them are listed in Table II.

2. PLANT DYNAMICS STUDIES

The following are some of the main tasks of the reactor plant dynamics experimental studies:

1. Determination of the maximum power the reactor may be stable at and the ways of raising it.

2. Determination of the reactor dynamic characteristics with the view of structural modifications to be carried out to improve them and the optimum automatic control system development.

3. Determination of various core physical characteristics such as control rod worth reactivity coefficient value etc.

4. Determination of optimum start-up and shut-down regimes (especially with steam superheating) as well as optimum scram system.

5. Verification of predicted dynamic equations. The plant equipment (steam generators, the turbine, etc.) dynamics as well as reactor dynamics will be also studied. Over a hundred special quick response sensors located in different plant parts are envisaged. For small thermocouple e.m.f. fluctuations special amplifiers will be used. These amplifiers provide an exact and stable compensation of the constant signal component and have some microvolt threshold of response.

The sensors of plant instrumentation and control devices are used for some measurements but in most of the measurements special sensors for dynamic studies are used.

MAIN CHARACTERISTICS TO BE MEASURED IN THE REACTOR VESSEL

Table II

No.	Characteristic	Sensor location
1	2	3
1	Neutron flux	Around the core
2	Power density distribution	Central tube in every assembly
3	Steam content	Upstream flow section, downstream flow section, in the core.
4	Water velocity	Downstream flow section at the core inlet; at the steam outlet from the superheater zone.
5	Pressure	Reactor vessel vapour plenum; superheated steam pipes.

1	2	3
6	Pressure drop	Core inlet — vessel vapour plenum, core outlet — vessel vapour plenum, superheated steam inlet and outlet in superheating fuel assemblies.
7	Temperature	Downstream flow section, fuel element cans in the boiling zone, fuel element cans in superheater zone.

The main reactor characteristics recorded during studies are listed in Table II.

On the plant as a whole and in its circuits steam and water flow rate, pressure and temperature in different pipes of the circuits are measured.

In addition to these characteristics all control mechanism positions are also indicated.

Oscillations will be caused by means of plant control mechanisms (control rods, main steam pipe throttle valves, circuit control valves, feed valves, turbine control valves, valves of steam dumping lines, etc). As most of control valves are provided with pneumatic drives, changing air pressure over the diaphragm may cause variations in parameters with the necessary amplitude at relatively high frequencies.

A special oscillator must be installed in the reactor to cause periodic reactivity oscillations with a frequency of 10 — 20 c.p.s.

The studies in the plant dynamics will be carried out by three techniques (5, 6).

The perturbation technique is the simplest from the point of view of data processing and it is most exact at the same time. Its disadvantages are long time necessary to carry out experiments and relative complexity of apparatus to cause the perturbations.

In addition, the use of the method is limited to investigations in linear approximation, that is under very small parameter deviations from the steady-state values.

A special infralow-frequency oscillator of sinusoid signals with electric and pneumatic outputs will be used to measure dynamic characteristics by the perturbation technique. It will make it possible to send a signal to the regulator demands as well as the control pneumatic valves directly.

Sending the signal to the control valves, it may be possible to cause oscillations of rather high frequencies (up to a frequency of 1 c.p.s.), this being of utmost importance when studying pressure influence on other reactor parameters. To remove non-linear distortions and noise input and output oscillations are expanded in Fourier series and amplitude and phase of the first signal harmonics are compared, processing of a great number of periods being required to remove noise influence. Analysis of frequency characteristics will be carried out automatically by means of a harmonic analyzer. This device is based on "synchronous demodulation" principle. Such devices are rather widely used to study any plant dynamics, in particular, a nuclear reactor dynamics (7). The peculiarity of the harmonic analyzer to be used for studies in the given plant dynamics consists in simultaneous analysis of six parameters. In addition, it is provided with input converters to analyse signals from various sensors. The analyzer is designed on the basis of AP-2 device (8).

Special apparatus to cause oscillations is not required, while using the method of non-periodic oscillations as it is sufficient to actuate a control mechanism remotely, somewhat displacing it. The input and output parameters are recorded. This method will be used to in-

ve stigate the processes with small oscillations in linear approximation as well as large variations in parameters (start-up and shut-down conditions). Quick response counters are provided to carry out experiments by this method. Experimental data processing will be carried out by means of digital computers.

The statistical method makes it possible to study a plant without any special oscillations. This is of a great importance while studying this plant as the operational conditions may restrict the possibilities of making oscillations. In addition, statistical investigations of the boiling reactor are of interest as the reactor power fluctuations are substantially affected by the stochastic character of the boiling process.

To study the statistical character of the reactor noise signals from the sensors are sent to the input unit of the correlatograph, recording twelve parameters simultaneously. The process records made on a magnetic tape are processed by the correlatograph computing device which results in determination of auto- and intercorrelation functions. The correlatograph has a wide frequency range (from 0.0001 to 20 c.p.s.). The input converters make it possible to record signals from the sensors of all kinds located at the plant. The subsequent analysis of the correlations functions by means of digital computers permits the transfer functions, transient functions, as well as the plant noise spectrum to be determined.

CONCLUSION

On the experimental power plant an extensive programme of studies in boiling reactor physics, in particular the reactor stability while in operation under different loads and pressures is to be carried out. The studies will permit to obtain data on the maximum permissible power level of the core. The operating experience of fuel elements at high heat fluxes will be very valuable for safety evaluation of fuel elements of various arrangements and compositions with cans of zirconium alloy under boiling conditions and of superheating fuel elements stability.

The economics of the power plant depends to a great degree on the fuel burn-up that will be determined during the test operation.

The problem of the neutron field distortion in the reactor will be also studied on the reactor under different conditions and different positions of the control mechanisms.

The reactor thermal and hydraulic characteristics will be studied too, in the first turn the problems of water circulation system safety and the influence of water and steam parameters and plant power on it.

A main problem of studies in the operating experience of the plant as a whole consists in verification of the turbine operating conditions when radioactive steam is fed to it and the choice of the best way to adjust the reactor power to the turbine one.

During the plant operation the requirements for water quality and the methods to maintain the purity required will be verified too.

Thus the test operation is to give experience necessary for the further development of nuclear power stations of different output with boiling reactors and boiling reactors with nuclear superheating of steam.

306

LIST OF REFERENCES

1. *V.S[illegible]lyanov*, Future of Nuclear Power Engineering in the U.S.S.R., Report No.2027 to the 2nd Geneva Conference on Peaceful Uses of Nuclear Energy, 1958.
2. 420 Mw Nuclear Power Station Design, U.S.S.R., Report to 11th Sectional meeting of the World Power Conference, Belgrade, 1957.
3. *E.I.Romm*, Boiler Plants, vol.2, 1946.
4. Directory of Nuclear Reactors, vol. 1, pp.73 – 76, International Atomic Energy Agency, Vienna,1959.
5. *E.P.Stefani*, Basic Principles of Regulator Setting of Heat-Power Processes Calculation, Gosenergoizdat, 1960.
6. *V.V.Solodovnikov* et al., Computers for Statistical Studies in Automatics, Mashgiz, 1963.
7. *G.Potye*, Device for Nuclear Reactor Transfer Function Measurements, Paper at 1st Congress of IFAC.
8. *V.G.Atamanenko*, Analyzer for Dynamic Characteristics Determination by Frequency Techniques, Priborostroenie, No.5, 1962.

306

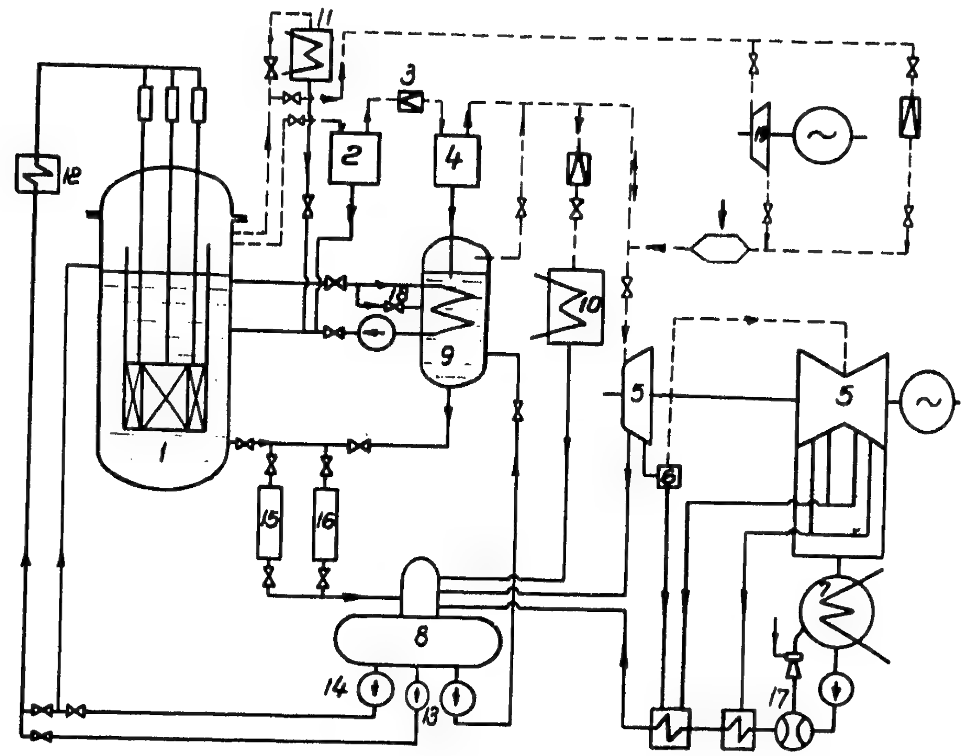

FIG.1. PRINCIPAL FLOW DIAGRAM OF THE PLANT

1 – reactor; 2 – high pressure separator; 3 – throttle valve; 4 – mean pressure separator; 5 – saturated steam turbine; 6 – moisture separator; 7 – turbine condenser; 8 – deaerator; 9 – steam generator; 10 – auxiliary condenser; 11 – shut down cooling condenser; 12 – heat exchanger; 13 – plunger pump; 14 – feed (water) pumps; 15 – water purifier evaporating unit; 16 – ion-exchange filter; 17 – turbine jet; 18 – steam generator feed line with step evaporation; 19 – superheated steam turbine.

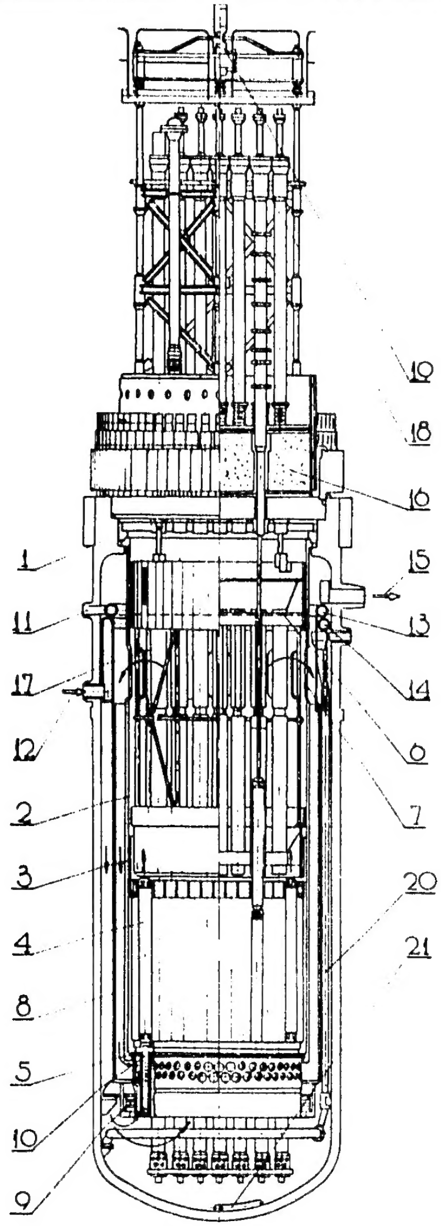

FIG.2. BOILING REACTOR GENERAL VIEW

1 – reactor vessel; 2 – chimney; 3 – basket; 4 – assemblies; 5 – lower grid; 6 – average water level; 7 – vessel penetrations; 8 – shield; 9 – gasket; 10 – sectional feed removable plugs; 11 – water outlet to the steam generators; 12 – water flow back from the steam generator; 13 – feed water distribution collector; 14 – water from the steam generator distribution collector; 15 – steam outlet; 16 – head; 17 – shielding tubes; 18 – control mechanism drives; 19 – traverse; 20 – liquid poison injection tube; 21 – drain tube.

306

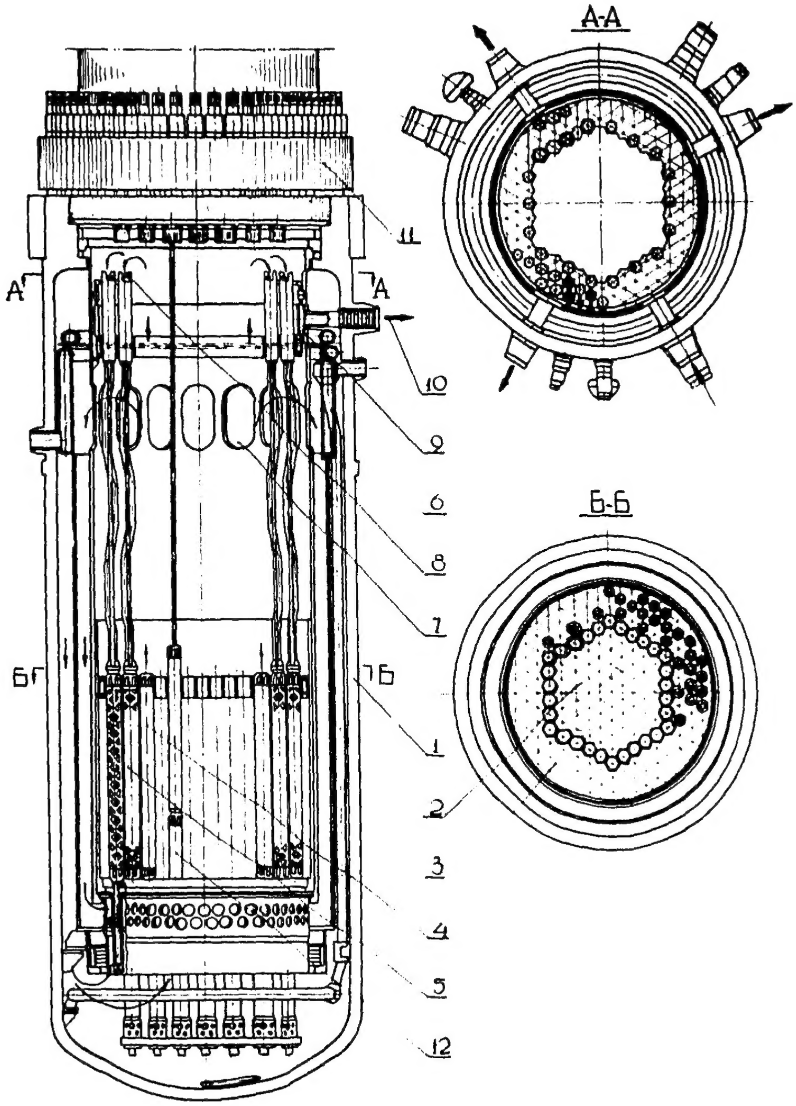

FIG.3. STEAM SUPERHEATING REACTOR GENERAL VIEW
1 – reactor vessel; 2 – boiling zone; 3 – superheater zone; 4 – boiling fuel assemblies; 5 - steam superheating channel; 6 – water level; 7 - vessel penetrations; 8 – saturated steam inlet; 9 – superheated steam collector; 10 – superheated steam outlet; 11 – head; 12 – control mechanisms.

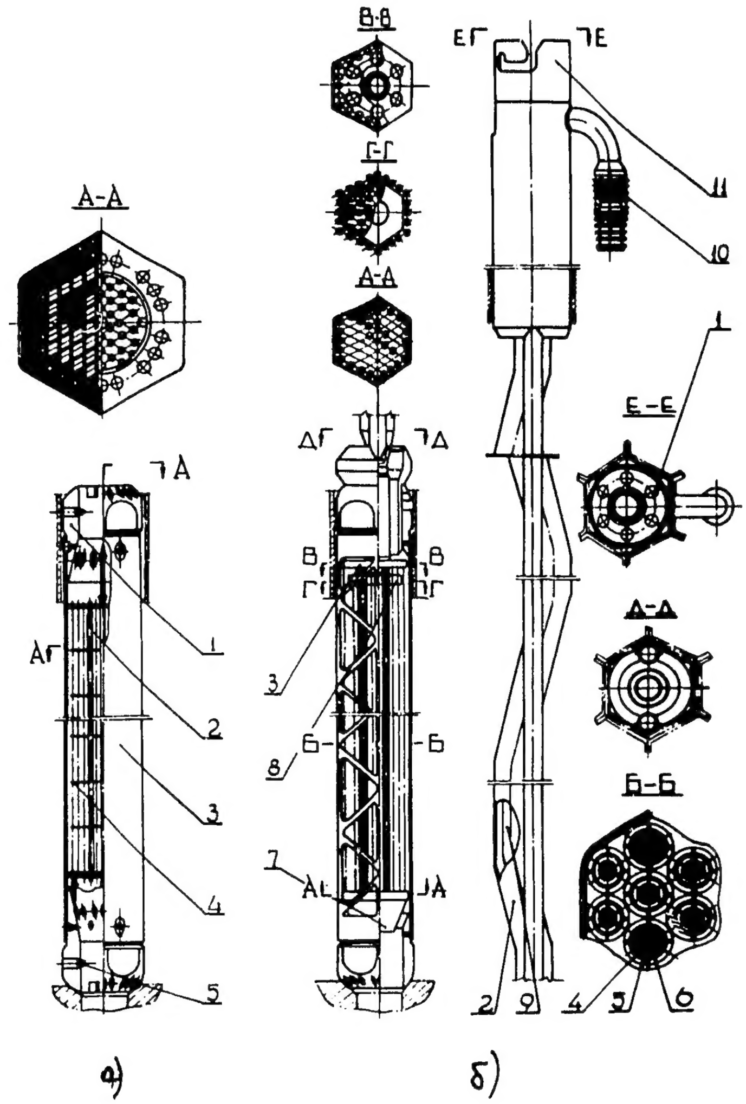

FIG.4. GENERAL VIEW OF THE BOILING ZONE FUEL ASSEMBLY AND SUPERHEATING STEAM CHANNEL

(a) Boiling zone fuel assembly; 1 – fuel assembly top; 2 – fuel elements; 3 – hexahedral sheath; 4 – spacer grid; 5 – rods to hold the assembly. (b) Steam superheating channel; 1 – saturated steam inlet penetration; 2 – saturated steam inlet; 3 – saturated steam collector; 4 – fuel element can; 5 – thermal insulation; 6 – fuel element; 7 – mixing collector; 8 – superheated steam collector; 9 – superheated steam outlet; 10 – sealing; 11 – grip.

306

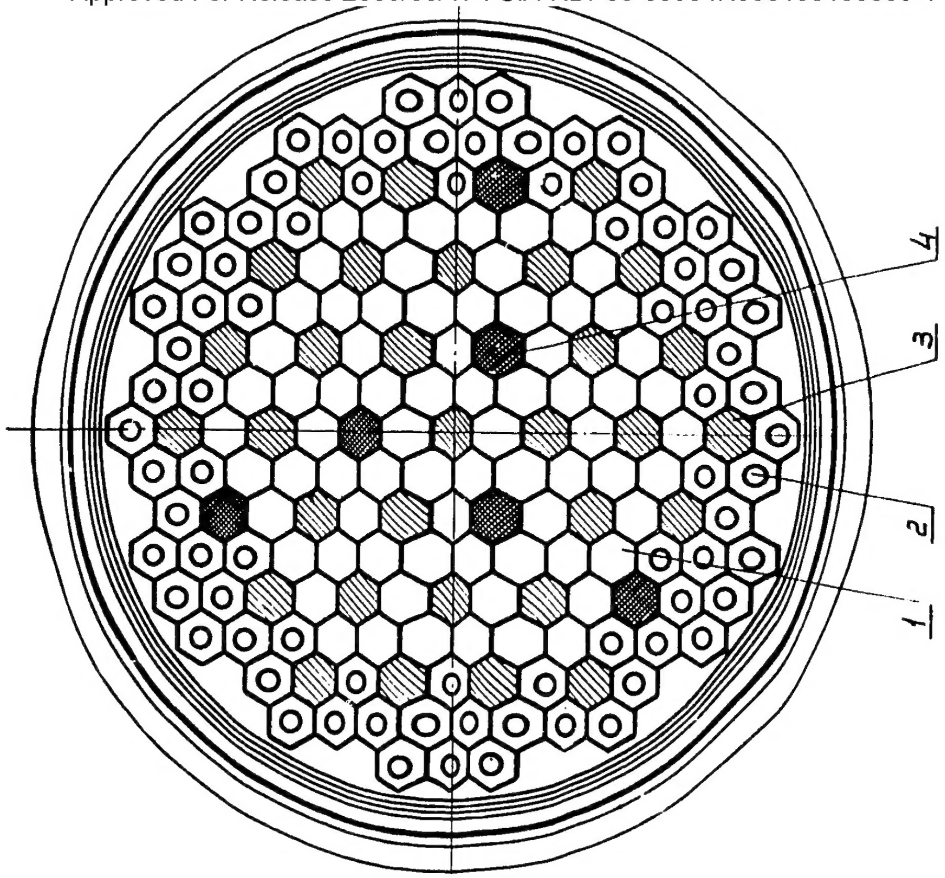

FIG.5. FUEL ASSEMBLY ARRANGEMENT IN THE CORE
1 – boiling zone; 2 – superheater zone; 3 – absorbers of excess reactivity compensation system; 4 – absorbers of safety system.

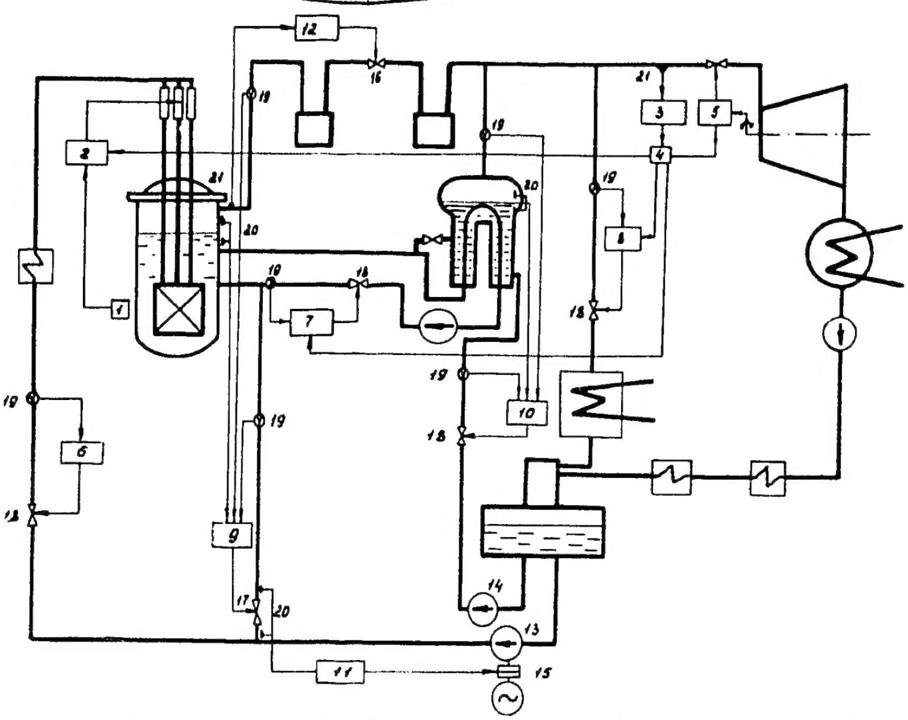

FIG.6. AUTOMATIC PLANT CONTROL SYSTEM (UNDER BOILING CONDITIONS)
1 – ion chambers; 2 – power regulator; 3 – command pressure regulator; 4 – switch; 5 – speed regulator; 6,.7, 8 – flow rate regulators; 9, 10 – feed water regulator; 11 – pressure drop regulator; 12 - pressure regulator; 13, 14 – feed pump; 15 – hydraulic clutch; 16, 17, 18 - control valves; 19 – flow rate sensor; 20 – level indicator; 21 – pressure sensor.

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100009-1